NON-PUBLIC?: N
ACCESSION #: 8804190038

LICENSEE EVENT REPORT (LER)

FACILITY NAME: Nine Mile Point Unit 2 PAGE: 1 of 8

DOCKET NUMBER: 05000410

TITLE: Reactor Scram and Emergency Core Cooling System Actuation due to a Loss of Feedwater Flow Caused by a Design Deficiency
EVENT DATE: 03/13/88 LER #: 88-014-00 REPORT DATE: 04/12/88

OPERATING MODE: 1 POWER LEVEL: 043

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Robert E. Jenkins, Assistant Supervisor Technical Support
TELEPHONE #: 315-349-4220

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: AD COMPONENT: PT MANUFACTURER: R369
REPORTABLE TO NPRDS: Y

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT: On March 13, 1988 at 17:39 with the reactor mode switch in Run (Operational Condition 1) and at a power level of approximately 43% (see note) rated thermal capacity, Nine Mile Point Unit 2 experienced an automatic reactor scram, an automatic initiation of the Division 3 Emergency Core Cooling System (ECCS) with a subsequent coolant injection, and the automatic actuation of several Engineered Safety Features. These events were the result of low water levels in the reactor vessel caused by a total loss of feedwater flow. An Unusual Event declared at 17:45 was terminated by 18:00 that day. The ECCS injection was manually terminated and a normal reactor shutdown was commenced by the NMP2 operators. (NOTE: Reactor power was at 98% two minutes prior to this event. However, an instrument failure caused the reactor recirculation pumps to downshift to low speed operation, decreasing reactor power to 43%.)

The immediate cause for this event is an equipment failure. However, the root cause for this event is a design deficiency.

The corrective actions for this event are; (1) a temporary modification has been performed to bypass the "seal in" logic for the low pressure heater string outlet isolation valves, (2) a permanent modification will be implemented to reroute the piping for the Second Point Feedwater Heaters level switches, and (3) the failed pressure transmitter, which was replaced with an upgraded model, will be sent to the vendor for a failure mode analysis.

(End of Abstract)

I. DESCRIPTION OF EVENT

On March 13, 1988 at 17:39 with the reactor mode switch in Run (Operational Condition 1) and at a power level of approximately 43% rated thermal capacity, Nine Mile Point Unit 2 (NMP2) experienced an automatic reactor scram, an automatic initiation of an Emergency Core Cooling System (ECCS), and the automatic actuation of several Engineered Safety Features (ESF). These events were the result of low water levels in the reactor vessel which were caused by a total loss of feedwater flow.

The sequence of events for this incident is as follows:

At 17:37:03, pressure transmitter 2ISC*PT122 failed, causing an erroneous low differential temperature signal for the steam dome/recirculation pump suction interlock. As a result, the Reactor Recirculation Pumps (RCP) automatically downshifted from high speed to low speed operation. Reactor power which was approximately 98% prior to this event decreased to approximately 43% within 15 seconds after the RCP downshift.

Between 17:37:38 and 17:39:00, the sharp reduction in reactor power caused a low pressure condition in the Extraction Steam System (ESS). Reduction in the extraction steam pressure caused steam flashing in the Second Point Feedwater Heaters (SPFH) affecting the level instrumentation for that equipment. This resulted in intermittent false high level signals for several of the low pressure feedwater heaters, which initiated an isolation for two of the low pressure feedwater heater strings. (These spurious signals were of an extremely short duration, typically alarming and clearing within one second. However, the outlet isolation valves (2CNM-MOV32A(B,C)) for the low pressure feedwater heater strings have a "seal in" feature requiring only a single instantaneous high level signal to close these valves.)

At 17:39:00, the "A" and "C" low pressure feedwater heater strings were completely isolated.

At 17:39:33, the Reactor Feed Pumps (RFP) tripped on a low suction pressure condition. This condition resulted from a reduction of flow to the RFP's after the "A" and "C" low pressure feedwater heater strings isolated. Tripping of the RFP's resulted in a loss of feedwater flow to the reactor.

At 17:39:38, the reactor water level began to decrease as a result of the loss of feedwater flow. At this time the reactor low water level (Level 4) alarm was annunciated in the NMP2 control room.

At 17:39:50, the reactor scrammed on a reactor low water level (Level 3) trip.

At 17:39:57, the NMP2 licensed operators placed the reactor mode switch to shutdown.

Due to the continued loss of reactor water inventory (due to boiling by decay heat), reactor water level reached the low-low level trip setpoint at 17:40:02. A reactor low-low water level (Level 2) trip signal was generated (as expected) which initiated the automatic actuation of the following systems:

1. The High Pressure Core Spray (HPCS) system
(Note: HPCS is a Division 3 ECCS system.)

2. The Division 3 Emergency Diesel Generator (EDG2)

3. The Reactor Core Isolation Cooling (RCIC) system

4. The Division 1 and Division 2 Standby Gas Treatment Systems (GTS)

5. The Division 1 and Division 2 Reactor Building Ventilation (HVR) Unit Coolers

6. Recirculation pump trip

7. Alternate Rod Insertion

8. The Division 1 Control Building Special Filter Ventilation (HVC) system

9. Isolation of the Normal Reactor Building Ventilation system

10. The Division 2 HVR Emergency Recirculation Unit Cooler (2HVR*UC413B)

11. Isolation of the primary containment except for the Main Steam Isolation Valves (MSIV's)
(Note: Primary Containment Isolation Valve Groups 4 and 5 isolate on a Level 3 signal.)

At 17:40:04, the HPCS system started to inject water from Condensate Storage Tank (CST) "B" to the reactor vessel. As a result, reactor water level began to increase.

At 17:40:05, EDG2 attained its normal operating parameters. Additionally, the ventilation system for EDG2 automatically started up. (The diesel generator ventilation systems are considered ESF systems.)

At 17:40:15, the RCIC system injected to the reactor from CST "A". As a result of the RCIC injection, a simultaneous Main Turbine-Generator trip was initiated.

At 17:42:27, the NMP2 operators restored partial feedwater flow to the reactor.

At 17:42:49, the reactor water level was restored to its normal level by the feedwater system and the HPCS and RCIC injections. All low water level annunciation had cleared by this time.

At 17:42:57, the HPCS injection was manually terminated by the NMP2 operators.

At 17:45, in accordance with Emergency Action Procedure EAP-2, an Unusual Event was declared for NMP2. Additionally, the NMP2 operators placed the Scram Discharge Volume (SDV) bypass switches to bypass. (This is done to prevent another scram signal on high SDV water level.)

At 17:48:03, the alternate rod insertion initiation was reset by the NMP2 operators.

At 17:51, the reactor scram was reset by NMP2 operators in accordance with Operating Procedure N2-OP-101C.

At 17:55, NMP2 operators secured the HPCS system.

At 18:00, the Unusual Event was terminated.

At 18:12, the primary containment valve group 6 and 7 isolations were bypassed by Operations in order to place the Reactor Water Cleanup (RWCU) system back into service.

At 18:42, the RCIC injection was secured by the NMP2 Operations Department.

Between 18:43 to 18:50 the Division 1 and Division 2 GTS systems were secured and normal HVR was restored by the NMP2 Operators.

At 19:56, the Main Turbine-Generator trip was reset by the NMP2 Operations Department.

At 23:35, the NMP2 operators secured the Control Building Special Filter Ventilation System.

The duration for this event, from the initial event transient (failure of pressure transmitter 2ISC*PT122) to the termination of the Unusual Event was approximately 23 minutes. It is estimated that approximately 11,500 gallons of water were injected into the reactor vessel by the HPCS system from CST "B". Additionally, it is estimated that the RCIC system supplied approximately 38,000 gallons of water from CST "A" to the reactor vessel. A small increase in reactor water conductivity (well within Technical Specification limits) was noted after this event.

The individual systems and components functioned as designed. There were no other inoperable systems which contributed to this event. No other plant system or component failure resulted from this event.

To satisfy the reportability requirement of TS Section 3.5.1(f) the following information is being provided for the ECCS high pressure coolant injection event:

Total accumulated initiation cycles for the HPCS system (from receipt of the NMP2 operating license up to and including the March 13, 1988 event) = 3

The usage factor value for the HPCS injection nozzle (as of March 13, 1988) remains significantly below 0.70.

II. CAUSE OF EVENT

The immediate cause for this event was the failure of pressure transmitter 2ISC*PT122. This instrument failure (which is considered to have been a random incident) generated a spurious signal which caused the RCP's to switch to low speed operation; this in turn caused the sharp reduction in reactor power. These initial events subsequently led to the reactor scram and the Division 3 ECCS actuation. However, this instrument failure is not considered to be the root cause for this event since the loss of feedwater flow (as it occurred in this event) is not an anticipated result for a transient involving a decrease in the reactor coolant system flow rate. (See the NMP2 Final Safety Analysis Report (FSAR) Section 15.3.)

Therefore, the most probable root cause for this event (and for the loss of feedwater flow) is a design deficiency. Spurious high water level signals, generated by the level switches for the SPFH's, initiated the "A" and "C" feedwater heater string isolations. These signals were the result of perturbations (caused by steam flashing) in the level switch instrument lines. (It is thought that the steam flashing (caused by the reduction in the ESS system pressure) may have forced water slugs through the sensing lines for the SPFH's level instrumentation.)

Niagara Mohawk Engineering has determined that the physical installation of the sensing lines for the SPFH level instrumentation made these instruments particularly susceptible to the transients caused by steam flashing. Engineering has concluded that a different installation configuration would make these level switches less susceptible to similar disturbances.

III. ANALYSIS OF EVENT

This event is considered reportable via 10CFR50.73(a)(2)(iv) because the reactor scram and the various safety system actuations (such as the automatic startup of the Division 3 ECCS and of the Division 1 and 2 GTS systems, the automatic alignment of HVR and HVC to their emergency modes, and the primary containment isolation (except for the MSIV's)) were automatic ESF actuations.

A reactor scram occurred on a Level 3 reactor low water level trip as a direct result of the loss of feedwater flow. A reactor scram is a conservative plant response which does not pose any safety consequences. The spectrum of events (which include the Division 3 ECCS and all the ESF actuations discussed above) that occurred as a result of the loss of feedwater flow are bounded within the analysis of the "Loss of Feedwater Flow" event discussed in the FSAR Section 15.2.7.

Reactor water level decreased slightly below the Level 2 trip setpoint. (Actually, the lowest water level attained was 100 inches above instrument zero.) HPCS and RCIC initiated as designed at the Level 2 setpoint and injected coolant from the CST's "B" and "A" respectively. Reactor water level was restored to its normal level approximately 3 minutes after the injection commenced. After the reactor water level was restored to normal, operators manually secured the HPCS and RCIC injections.

The temperature difference between the coolant injected into the reactor vessel and the reactor coolant was approximately 460 degrees. However, Niagara Mohawk Engineering has determined that the injection did
not
cause a damaging transient to the reactor components.

The automatic actuation of the HPCS system with a subsequent coolant injection was a conservative plant response with minimal plant impact and no resultant impact on public safety. The HPCS actuation is considered conservative because the ECCS systems are designed to provide timely protection against onset and consequences of conditions that threaten the integrity of the fuel barrier and the Reactor Coolant Pressure Boundary.

The other ESF actuations (i.e., Division 1 and 2 GTS startup, lineup of the HVR and the HVC systems to their emergency modes, and the primary containment isolation) were also conservative plant responses, with minimal plant impact and no resultant impact on public safety.

The primary containment valve isolation is considered conservative since the primary objective of the isolation function is to provide protection to the plant and public by preventing releases of radioactive materials to the environment.

The GTS system is designed; (1) to limit the release of radioactive gases from the RB to the environment within the guidelines of 10CFR100 in the event of a loss of coolant accident and, (2) to maintain a negative pressure in the RB under accident conditions. (The emergency recirculation mode of HVR helps achieve these objectives.) Therefore, an automatic initiation of GTS and the emergency recirculation mode of HVR are considered conservative since their proper function serves to limit and contain radioactive releases from the primary and secondary containments.

Finally, the automatic alignment of the HVC system to the special filter ventilation mode provides monitored and filtered air to the control room under accident conditions. This mode of HVC minimizes the influx of radioactive contaminants into the control room. Therefore, an automatic alignment of HVC to this mode of operation is a conservative response.

The elapsed time for the event, from the initial event transient to the termination of the Unusual Event was approximately 23 minutes.

IV. CORRECTIVE ACTIONS

(1) To minimize future losses of feedwater flow due to spurious feedwater heater string isolations, a temporary modification (#88-104) has been implemented to bypass the "seal in" feature for the low pressure feedwater heater string outlet isolation valves (2CNM-MOV32A(B,C)). (The decision to remove or to permanently implement this modification will be based upon the operating experience of NMP2.)

(2) As an effort to desensitize the SPFH level transmitters to short term perturbations such as those caused by steam flashing, Modification (PN2Y88MX055) will be performed rerouting the piping for these instruments. It is anticipated that this modification will be installed no later than the mid-cycle outage (scheduled for September, 1988).

(3) The failed pressure transmitter(2ISC*PT122), a Rosemount Model #1152GP, was replaced with a Rosemount Model #1153GB via Work Request (WR #138201). This defective transmitter will be sent to the manufacturer for an analysis of its failure mode.

V. ADDITIONAL INFORMATION

LER's 88-01 and 88-12 also discuss events where the reactor scrammed on a low water level (Level 3) trip and the Division 3 ECCS system actuated. However, the causes for those events are not similar to the event discussed in this report. Therefore, there are no previous events similar to that discussed in this LER.

Failed Component Identification: Pressure Transmitter Model #1152GP8
Manufacturer - Rosemount
Vendor - General Electric (GE)
GE Part Number - 169C8393P882203

V. ADDITIONAL INFORMATION (Cont'd)

Identification of Components Referred to in this LER

IEEE 803 IEEE 805
Component EIIS Funct System ID

Pressure Transmitter PT AD
Level Transmitter LT SJ
Reactor Recirculation Pump P AD

Feedwater Pump P SJ
Low Pressure Feedwater Heater (SPFH) HX SJ
Piping (Tubing) TBG SJ
Diesel Generator DG EK
Scram Bypass Switches HS JC
Scram Discharge Volume COL AA
Isolation Valves (Primary Containment
& Feedwater Heaters) ISV JM,SJ
Main Steam Isolation Valves (MSIV's) ISV JM
HVR Unit Coolers CLR VA
Condensate Storage Tank TK KA
Reactor Mode Switch HS JC
Extraction Steam System N/A SE
Feedwater System N/A SJ
High Pressure Core Spray System N/A BG
Reactor Core Isolation Cooling System N/A BN
Standby Gas Treatment System N/A BH
Reactor Building Ventilation System N/A VA
Reactor Water Cleanup System N/A CE
Emergency Diesel Generator
Ventilation System N/A VJ
Control Building Ventilation System N/A VI
Primary Containment N/A NH
Reactor Recirculation System N/A AD

ATTACHMENT # 1 TO ANO # 8804190038 PAGE: 1 of 1

NIAGARA
MOHAWK NMP32818

NIAGARA MOHAWK POWER CORPORATION
301 PLAINFIELD ROAD, SYRACUSE, N.Y. 13212/TELEPHONE (315)474-1511

April 12, 1988

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Re: Docket No. 50-410
LER 88-14

Gentlemen:

In accordance with 10 CFR 50.73, we hereby submit the following Licensee
Event Report:

LER 88-14 Is being submitted in accordance with 10 CFR 50.73 (a)(2)(iv), "Any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS)."

A 10CFR50.72 (b)(2)(ii) report was made at 1810 hours on March 13, 1988.

This report was completed in the format designated in NUREG-1022, Supplement 2, dated September 1985.

Very truly yours,
/s/ Thomas J. Perkins
Thomas J. Perkins
Vice President - Nuclear

TJP/POB/mjd
Attachments
cc: Regional Administrator, Region I
Sr. Resident Inspector, W. A. Cook

*** END OF DOCUMENT ***